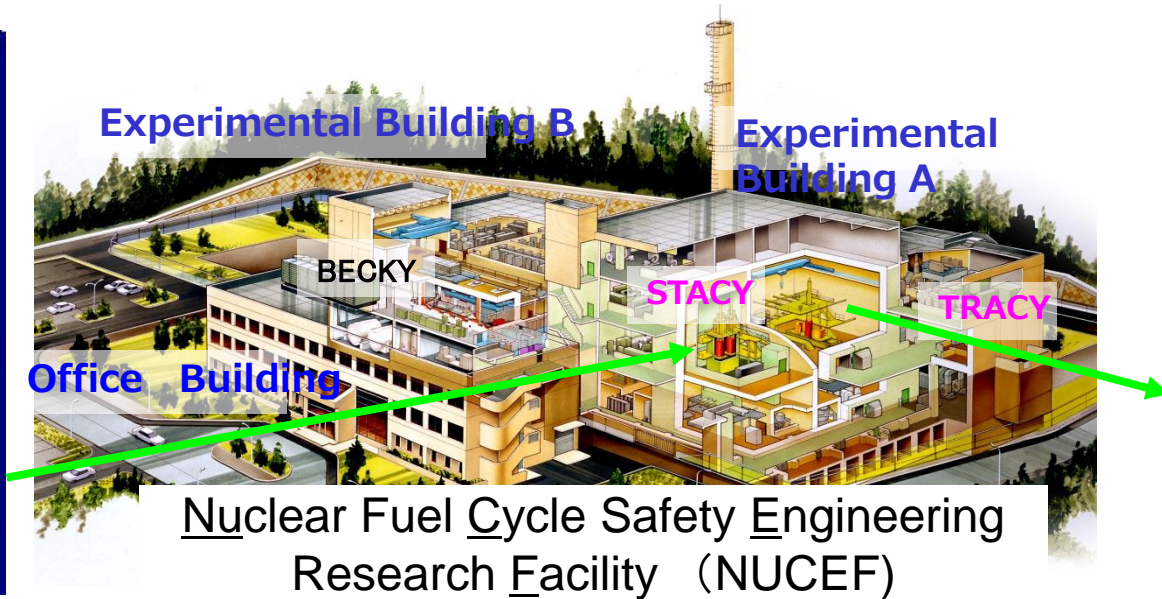


# JAEA's Effort and New Program for Criticality Safety of Nuclear Fuel Facility

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Japan Atomic Energy Agency

- Criticality safety research for nuclear fuel cycle facilities such as reprocessing and MOX fuel fabrication has been conducted to accumulate critical data by experiments and to improve evaluation methods for insuring the criticality safety of facilities.
- **Criticality experiment**
  - Research on static and kinetic feature of fuel solution system
  - Criticality experiments using STACY  
sub-criticality measurement
  - Criticality experiments using TRACY  
code development for criticality accident
- **Criticality evaluation**
  - Confirmation of criticality safety margin
  - Criticality safety handbook, evaluation of criticality
  - Developments of criticality evaluation codes and libraries for burn-up calculation, nuclear data issues



## STACY Experiments

Accumulation of critical data of low enriched U and Pu nitrate solution

- Basic homogeneous core tank
- Multiple unit core tank
- Heterogeneous core tank
- Temperature coefficient
- Kinetic Parameter
- FP effect, Absorber

## TRACY Experiments

▪ Accumulation of data on transient behavior of criticality accident

- Power, Temperature, Pressure
- Feedback mechanism (Temperature, Void effect)
- Spatial distribution of neutron, gamma ray
- Shielding effect

## Measurements

### Critical Mass

Critical solution level measurement  
with a high precision gauge,  
Dimension and composition of structure,  
Composition of solution fuel, etc.

### Temperature reactivity coefficient

Critical mass variation depending on  
solution fuel temperature (40°C max.)

### Kinetic parameter ( $\beta/\Lambda$ )

Pulsed neutron source method  
Reactor noise method

## Purposes

Criticality benchmark  
data

Provided for ICSBEP  
Code validation

Reactivity effect  
tabulation

Burn-up credit introduction

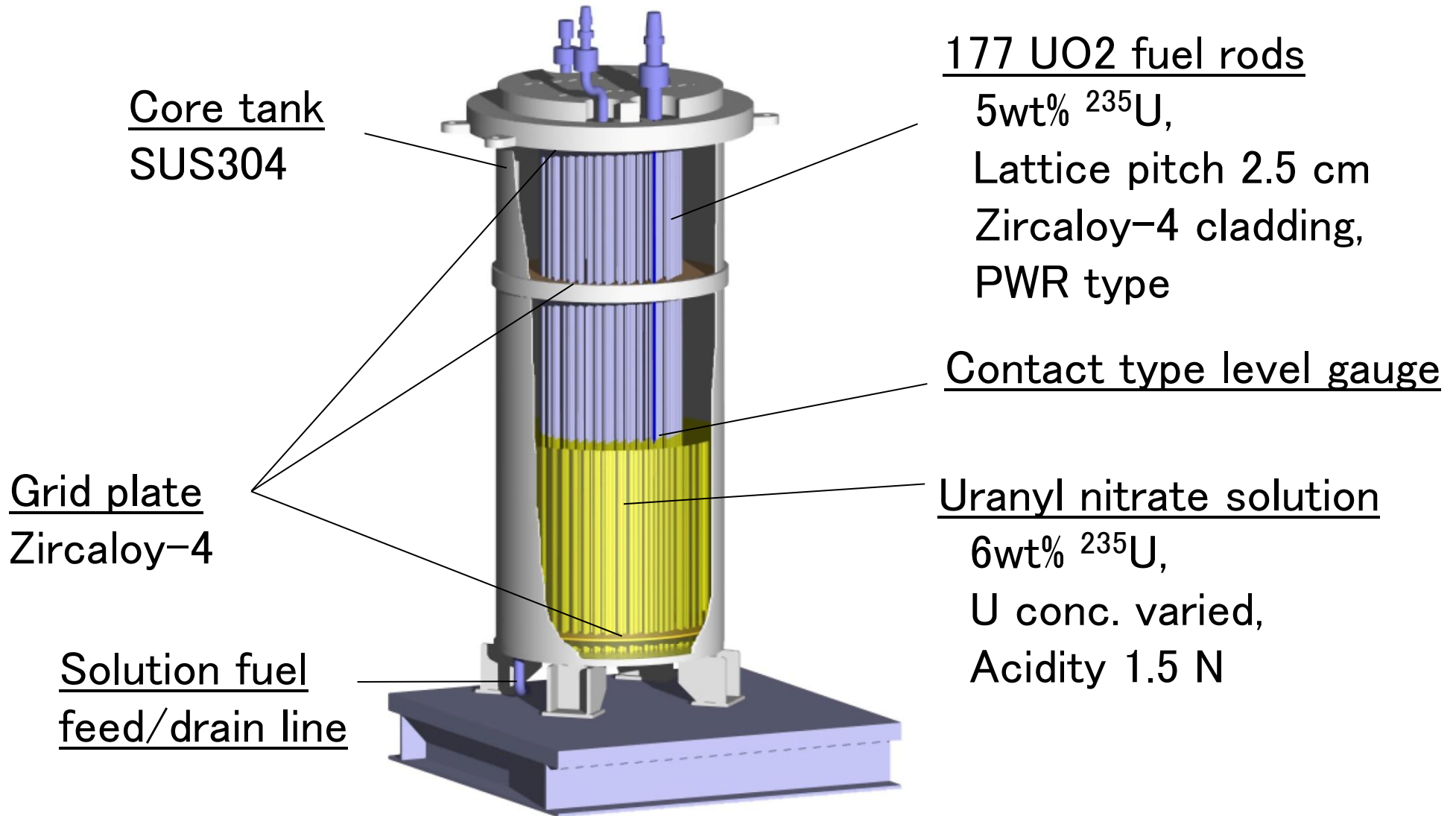
Basic data for criticality  
accident evaluation

Development of subcriticality  
monitoring techniques

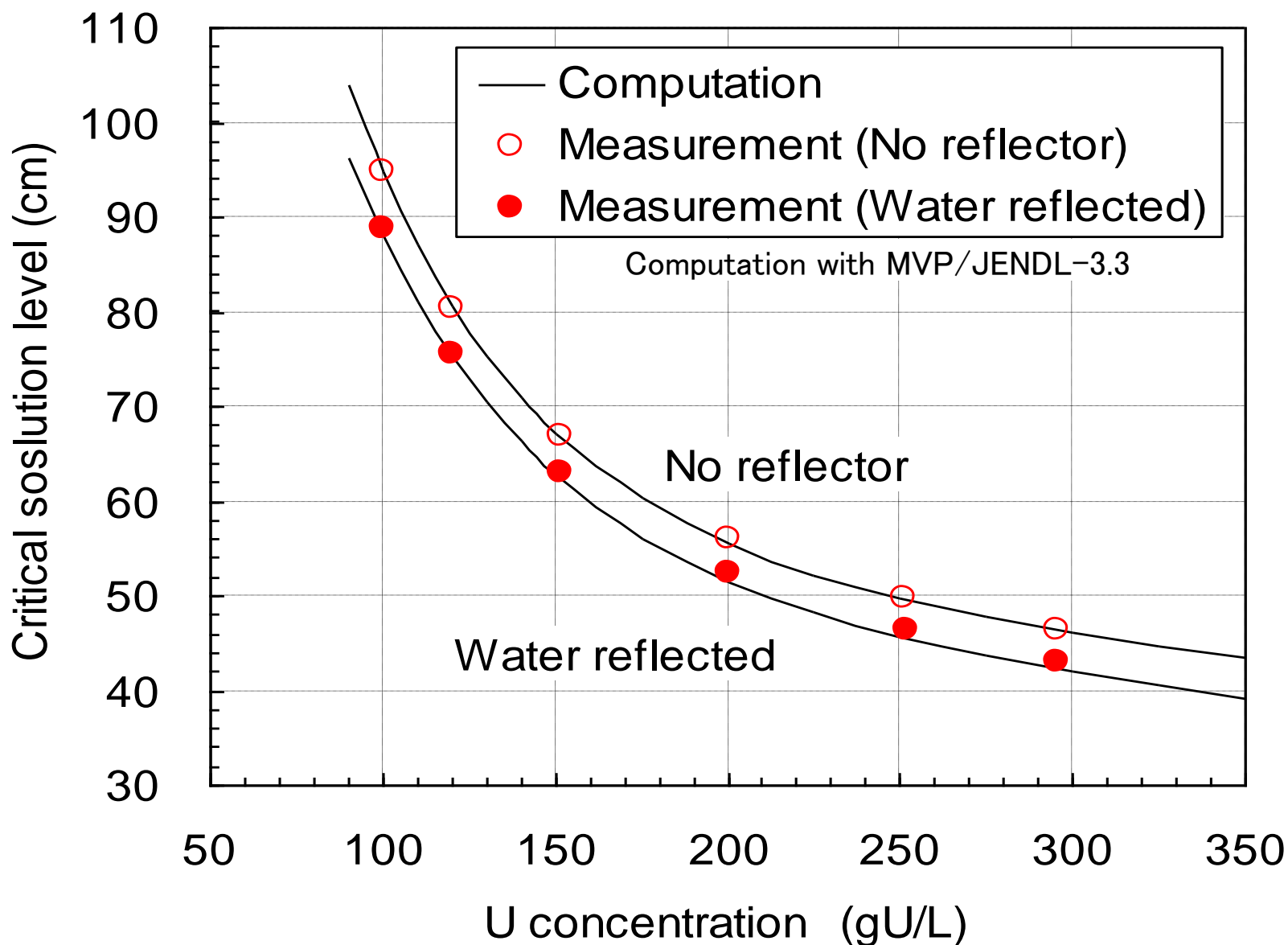
- 
- 1995 Single-unit, homogeneous system, 10%  $^{235}\text{U}$   
Cylindrical tank (600mm, 800mm Dia.),  
Slab tank (280mm × 700mm),  
Reflectors of various materials
  - 1999 Two-unit, homogeneous system, 10%  $^{235}\text{U}$   
Slab tanks (350mm × 700mm),  
Isolators of polyethylene and concrete
  - 2002 Heterogeneous (Fuel solution & rods) system  
(2.1cm pitch)
  - 2003 Heterogeneous system (1.5cm pitch)
  - 2004 Single-unit, homogeneous system, 6%  $^{235}\text{U}$   
Cylindrical tank (800mm Dia.)
  - 2005 Heterogeneous system with FP (1.5cm pitch)
  - 2006 Heterogeneous system with Gd (1.5cm pitch)
  - 2007 Heterogeneous system (2.5cm pitch)
  - 2008 Heterogeneous system (2.5cm pitch) .

# Heterogeneous Core System

## - Configuration -

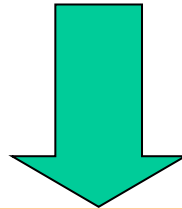


## - Results -



## Measurements

- Systematic accumulation of data on transient behavior at criticality accident conditions
  - Power, temperature, pressure v.s. excess reactivity, reactivity insertion ratio, initial reactivity, initial power, etc.

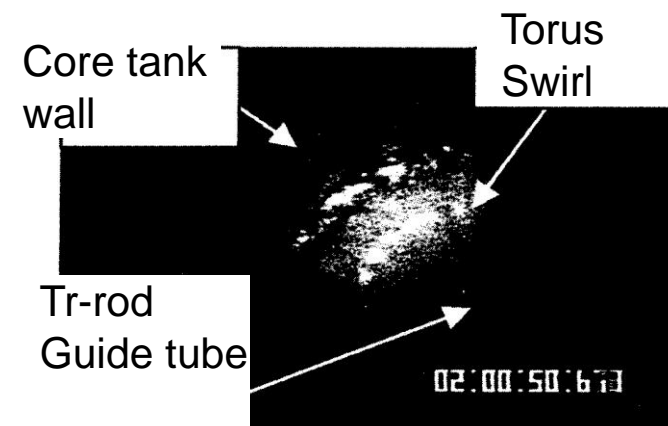


## Purposes

- Understanding phenomenon at criticality accident
- Development of criticality accident evaluation code
- Development of criticality accident prevention method

- 1995 First critical
- 96-98 Pulse Withdrawal, Ramp Feed, Ramp Withdrawal
- 1999 Cooperation with NIRS, Visualization
- 2000 Long duration(5h)
- 2001 Pressure measurement
- 2002 Reactivity temperature Coeff. measurement
- 2003-08 Water reflected core
- 2006-08 Initial Temperature Effect

NIRS: National Inst. Of Radiological Sci.

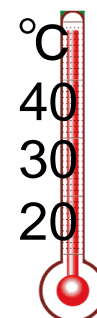


Visualization in core



Water reflected core

Ini. Temp. 23-33 °C



# Outline of TRACY

9

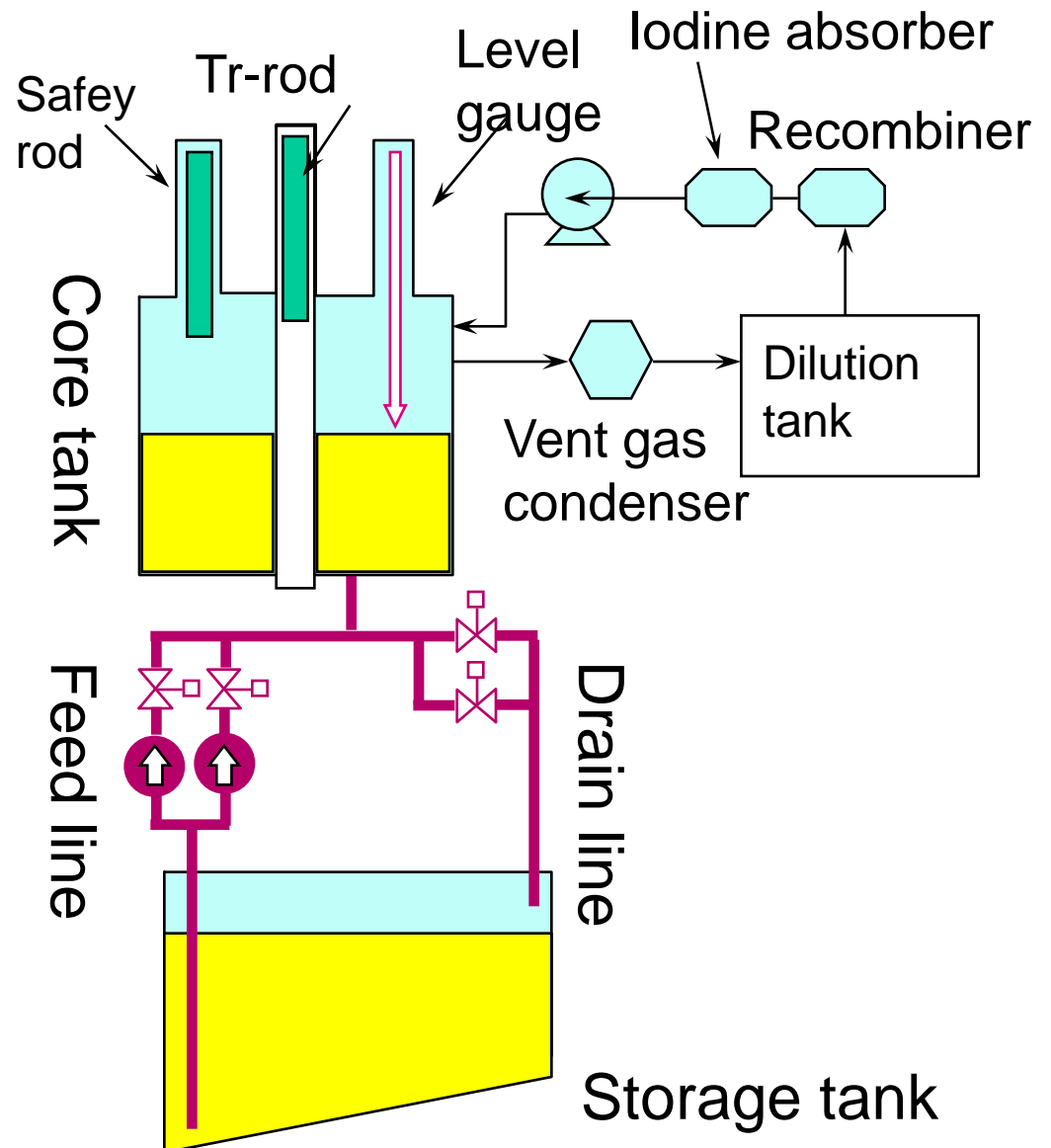
Core : annular  
(OD:50cm、ID:7.6cm)  
Fuel : Uranyl nitrate sol.  
(10%-enriched Uranium)

Max.Power : 5000MW  
Max.Energy:32MW·s/  
experiment  
( $\simeq 10^{18}$ fission)

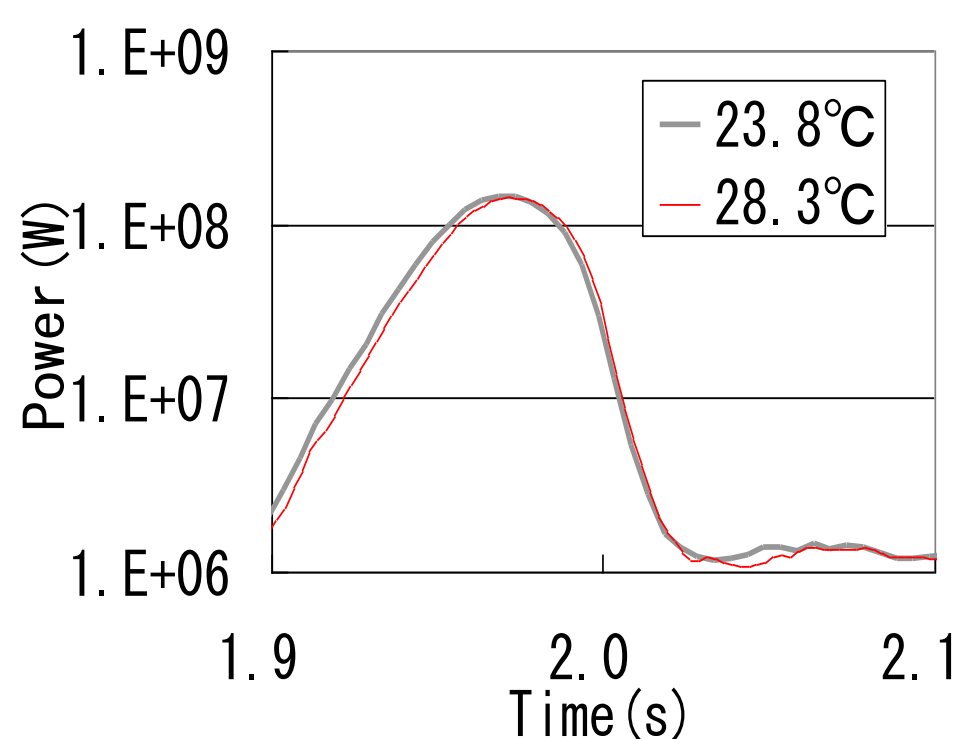
Max.Reactivity : 3 \$

Reactivity insertion method

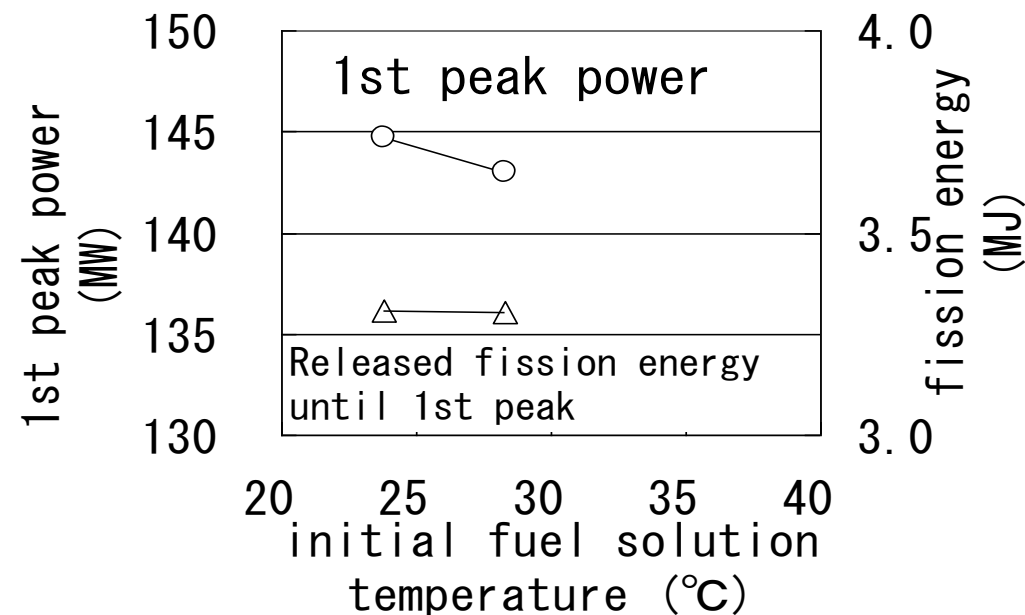
- ◆ RF: Ramp Feeding of fuel
- ◆ Withdrawal of Tr-rod
  - PW: Pulse Withdrawal
  - RF: Ramp Withdrawal



- Initial temperature effect measurements
  - Initial Temp. set from 23 to 33 °C
  - 2.5\$ inserted by Ramp Withdrawal of Tr-rod, 875cm/min.

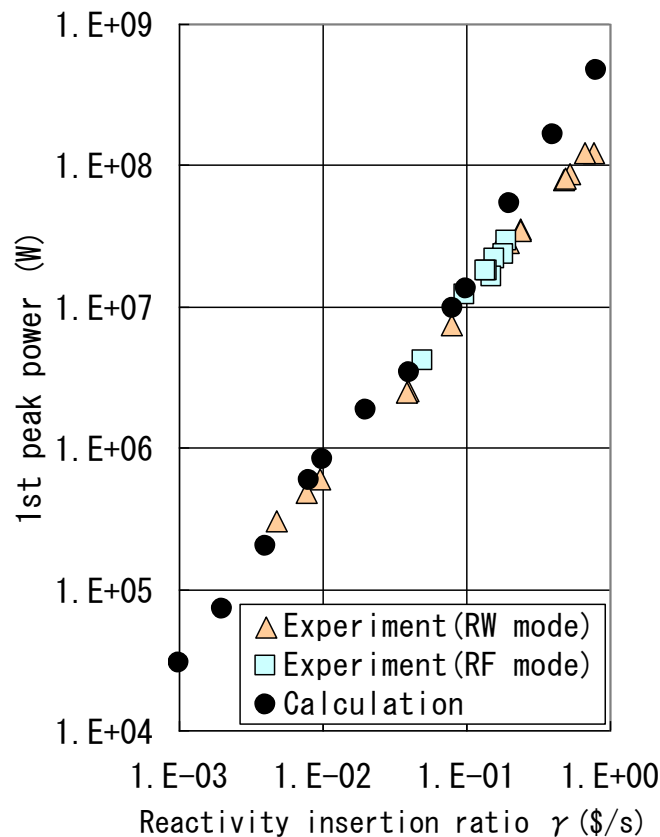


Power profile is almost same

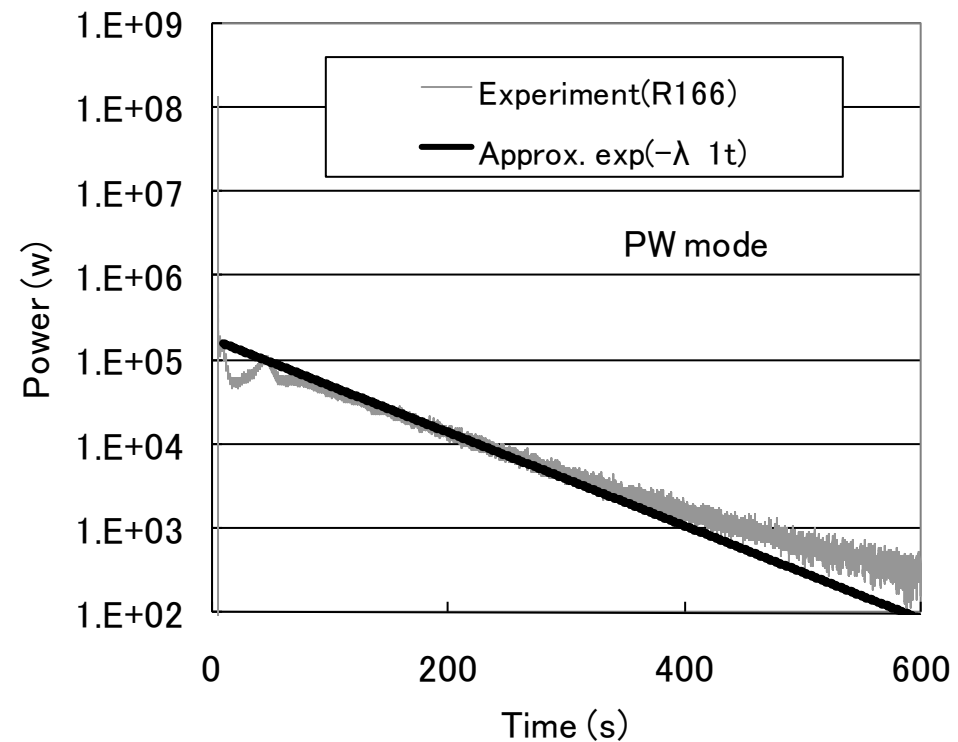


1<sup>st</sup> peak power is lower for higher initial temp.

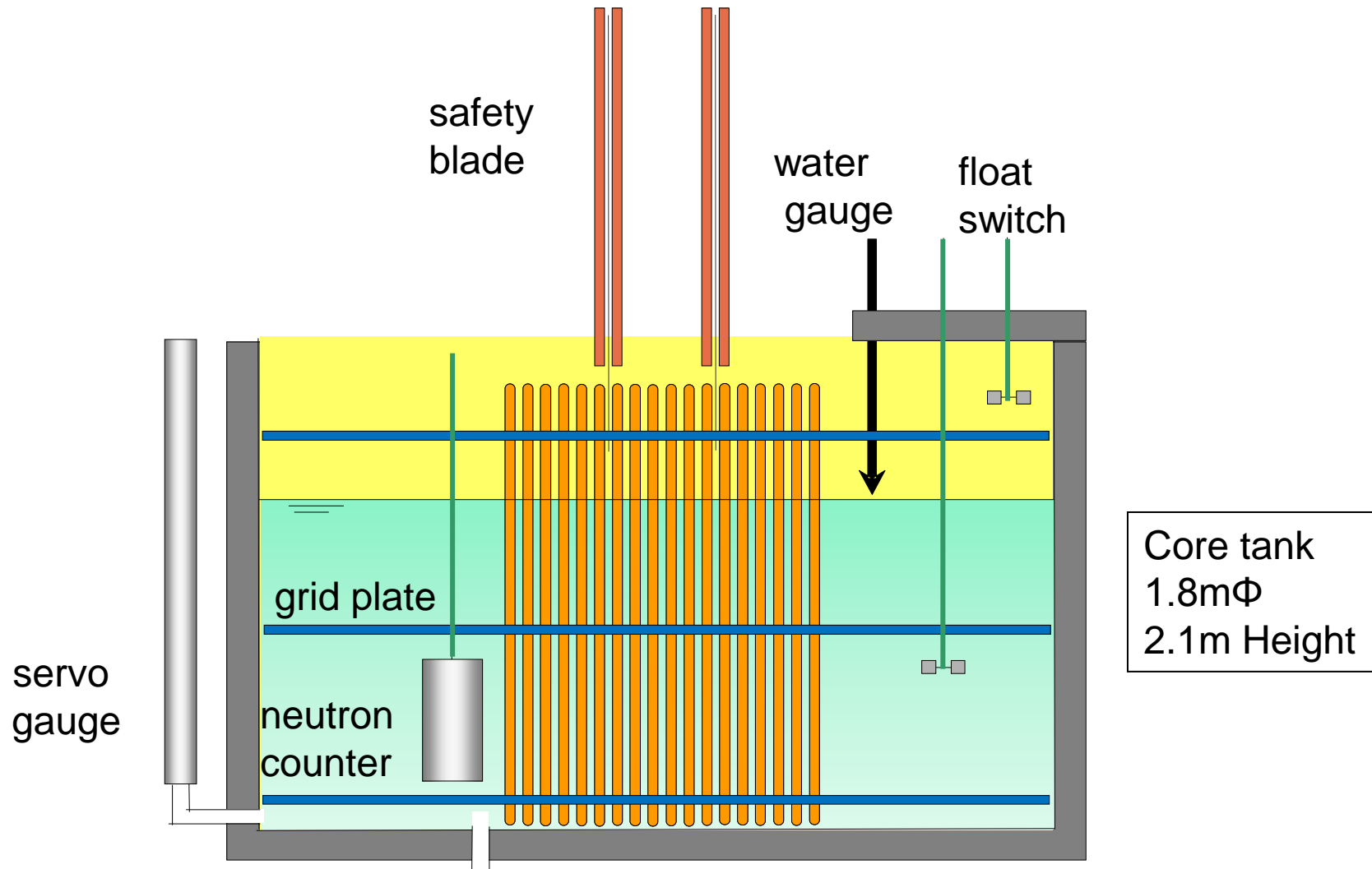
- Reproduction of a part of power profile
  - To understand the behavior of power at criticality accident
  - To achieve a complete simple method for fission yield



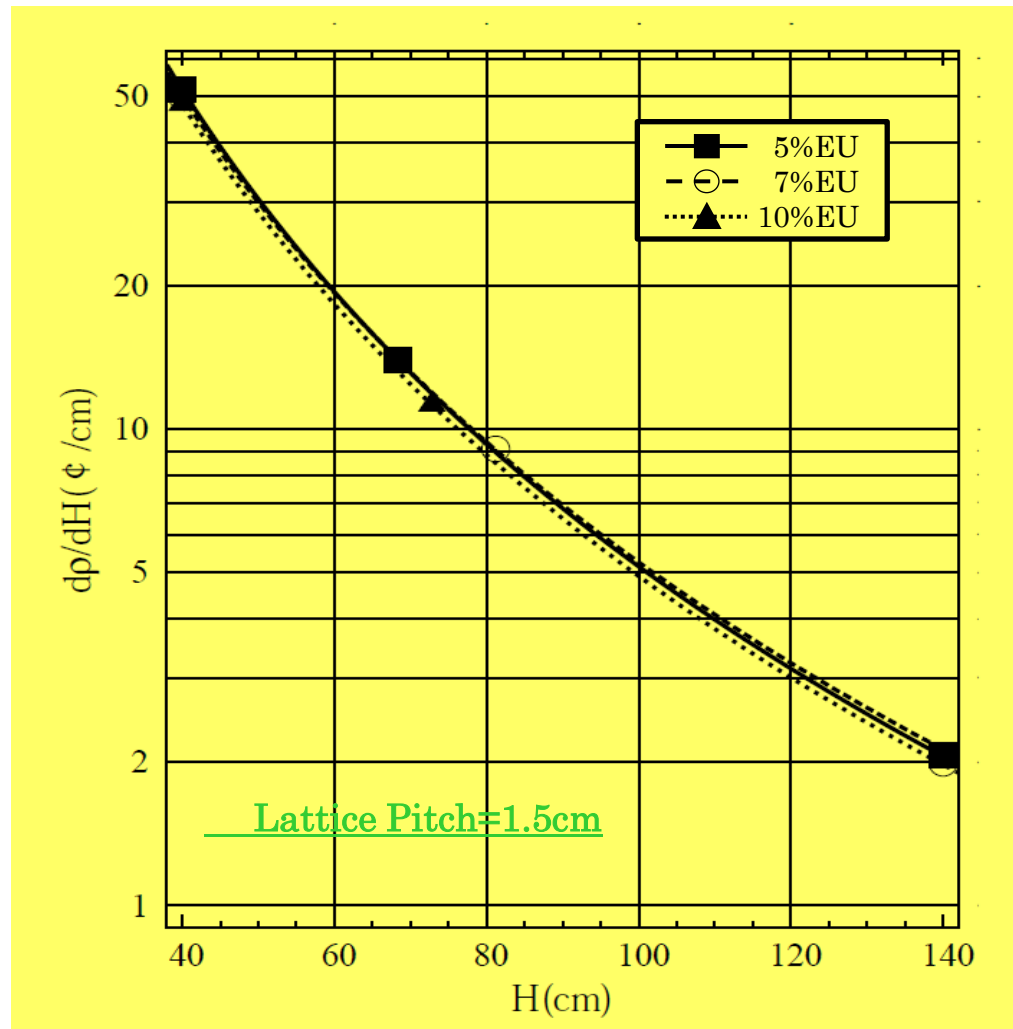
Peak Power of Slow Transient



Power Profile after 1<sup>st</sup> peak power

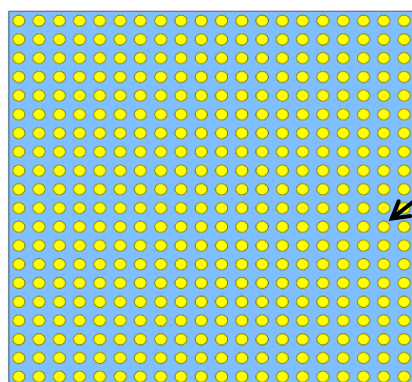


# Reactivity Control with Water Level



Water-level Worth

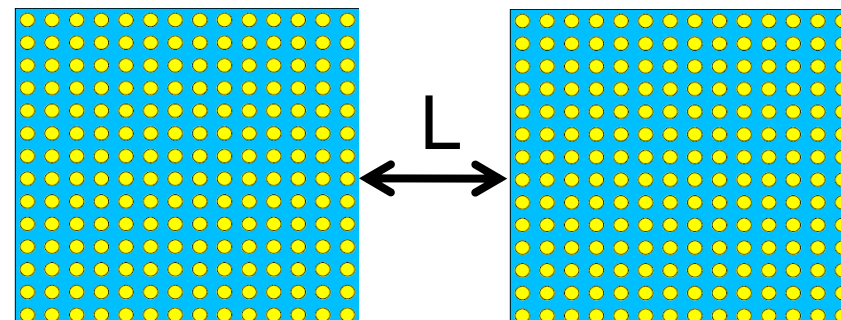
# Typical Core Configuration



fuel rod  
5~10 wt%

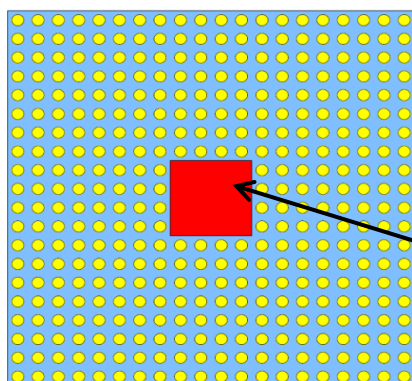
Base Core

(LWR lattice Pitch, Enrichment)



Two-interacting Cores

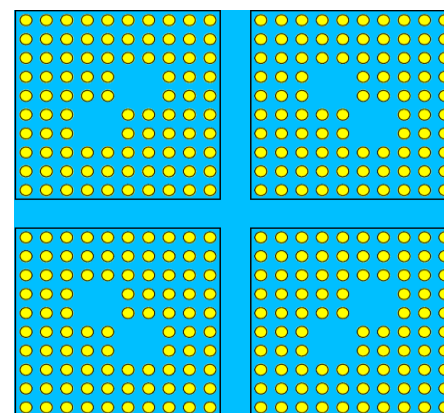
(Isolation, Structural material)



test region

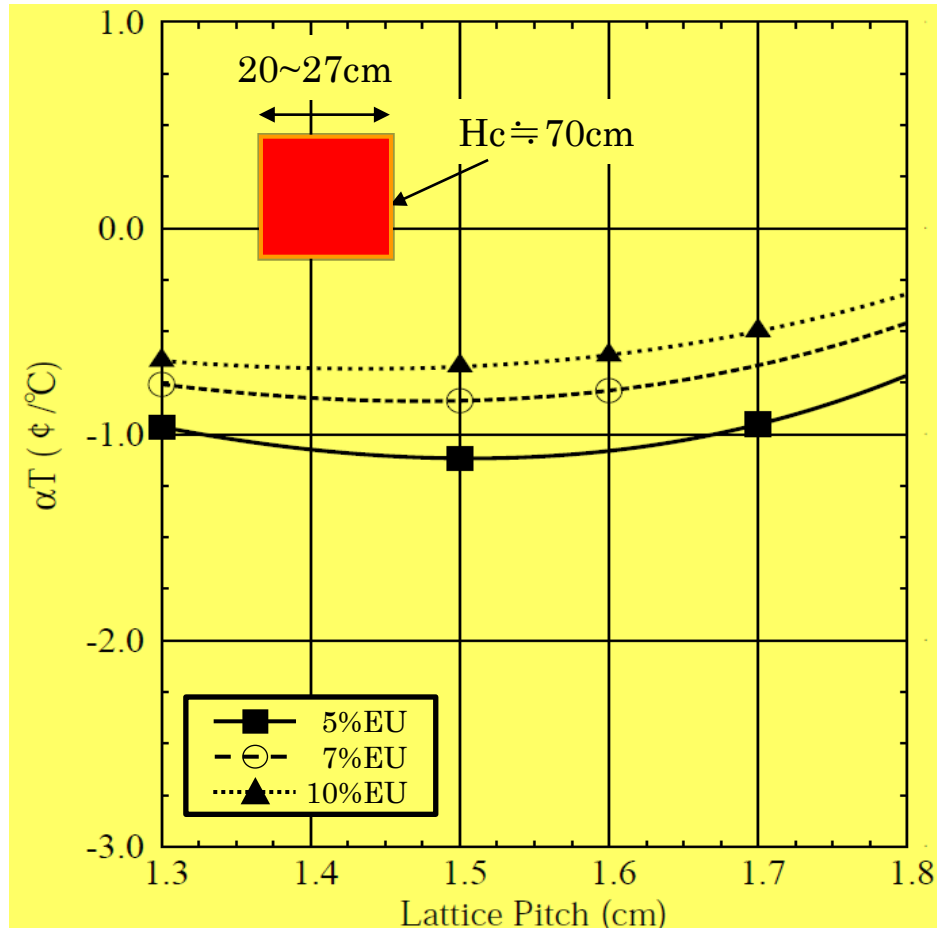
Two-region Core

(Sample worth, Solution, Powder )

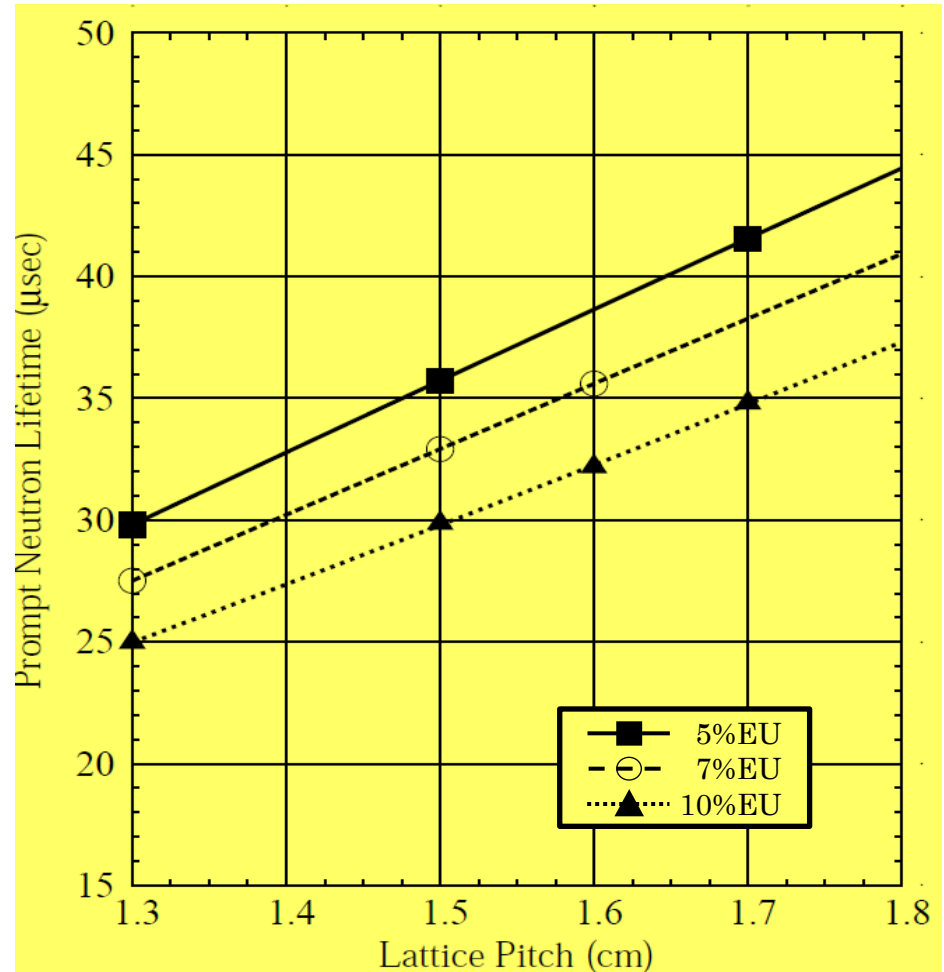


Multiple Core

(Storage, Assembly, )



Temperature\_coefficient



Prompt neutron life time

# Future Researches

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Criticality safety research for future fuel cycle facilities with innovative technologies

- **Criticality experiments**

- Research on static and kinetic feature of new kind fuels, in LWR cycle of the next generation, high burn-up, higher  $^{235}\text{U}$  initial enrichment( above 5%)

- **Criticality evaluation**

- Criticality safety handbook, evaluation of criticality
- Developments of criticality evaluation codes and libraries for burn-up calculation